Nuclear Power Safety:
Advanced Reactors and Key Issues

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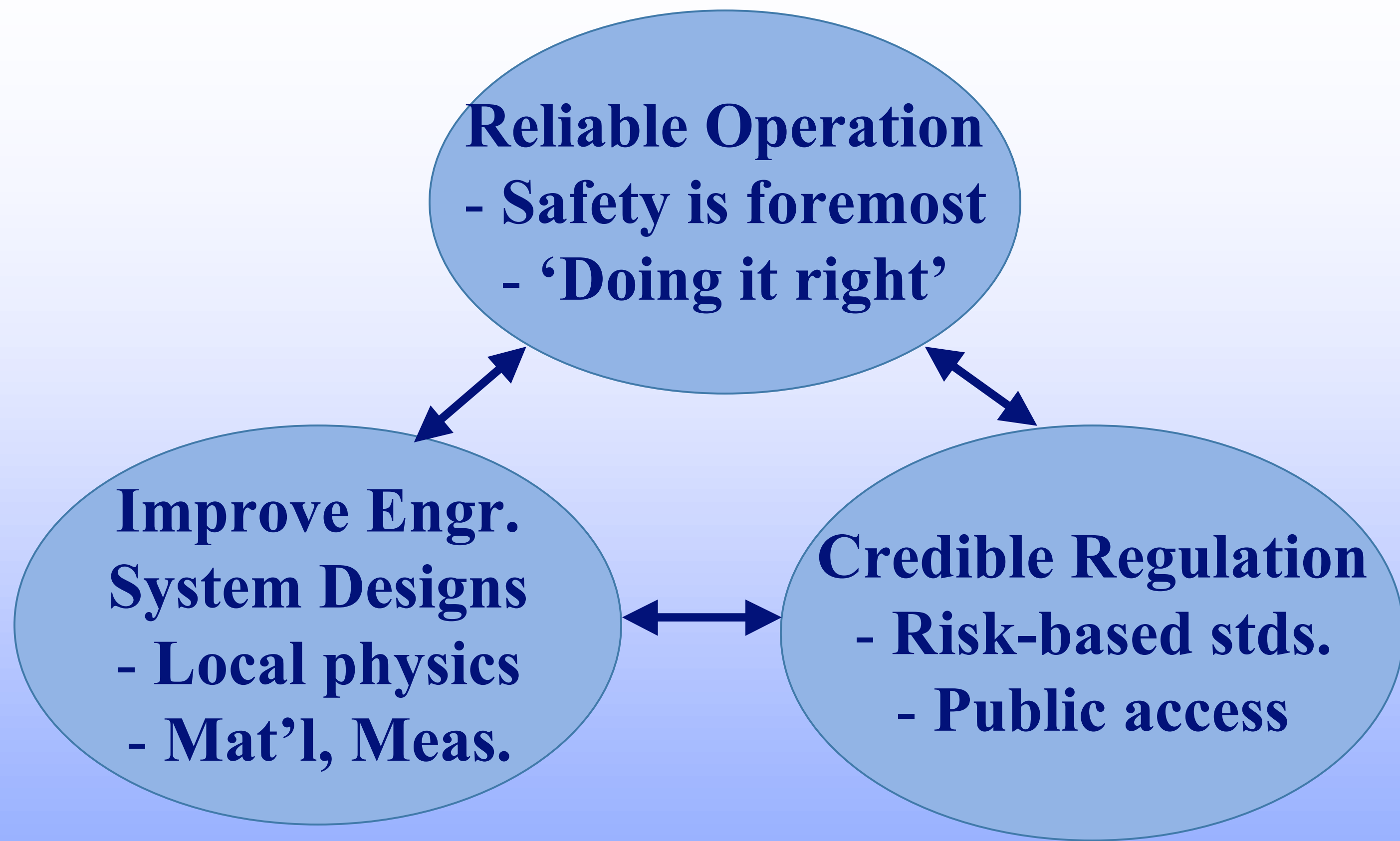
Concept of Engineering Safety

- Engineers consider safety integral to system design
- Engineering systems have a number of safety levels:
  - Engineering system should imbed safety in the design
  - System operation strives for high reliability
  - An engineering system designs for off-normal events
  - Robust engineering systems consider rare events
- Nuclear power safety => Avoid, minimize & mitigate the release of radioactivity: Defense-in-depth
  - Reliable operation, anticipate accidents, continual improvements in operator and systems performance

Nuclear Energy: Defense-in-Depth



Provide key info and enough time to make correct decisions

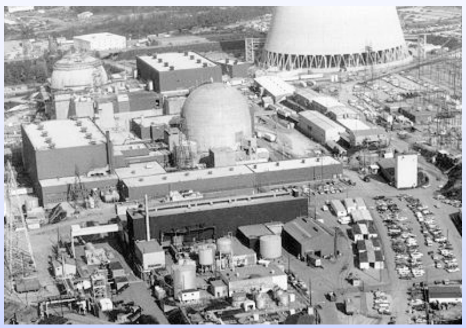
Nuclear Power Plant Safety

- There has been an impeccable safety record for nuclear power in the U.S. (no loss of life from commercial operation)
- Current LWR design demonstrates a high degree of safety to remove decay heat & minimize radioactivity release (e.g, TMI)
- Chernobyl accident was a terrible accident (negligent actions with a flawed engineering design: redesigned and retrained)
- More than two decades, safety focus is on best-estimates for Design-base events and Risk-informed estimates with PRA for events that may be judged beyond the design base
- This talk focuses on advanced reactors (fuel-cycles next):
  - Design-base events & associated safety issues
  - Beyond the design-base events and risk issues
  - Key issues and needs identified for Hi-Perf. Computing

Evolution of Nuclear Power Systems

Generation I

Early Prototype Reactors



- Shippingport
- Dresden,Fermi-I
- Magnox

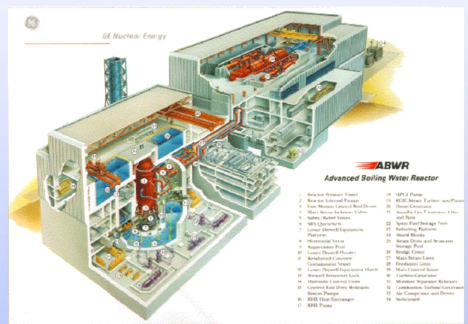
Generation II

Commercial Power Reactors

- LWR: PWR/BWR
- CANDU
- VVER/RBMK

Generation III

Advanced LWRs



- System 80+
- ABWR
- AP600
- SBWR

Generation IV

- Enhanced Safety
- Minimized Wastes
- Proliferation Resistance
- Highly economical

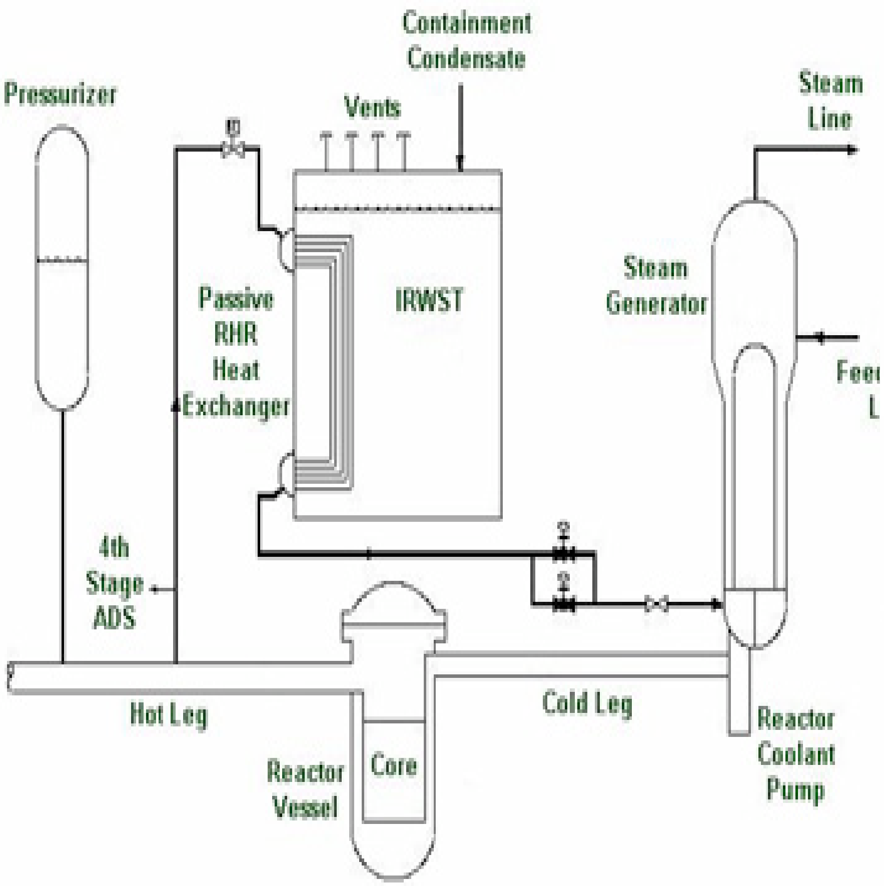
Gen I | Gen II | Gen III | Gen IV

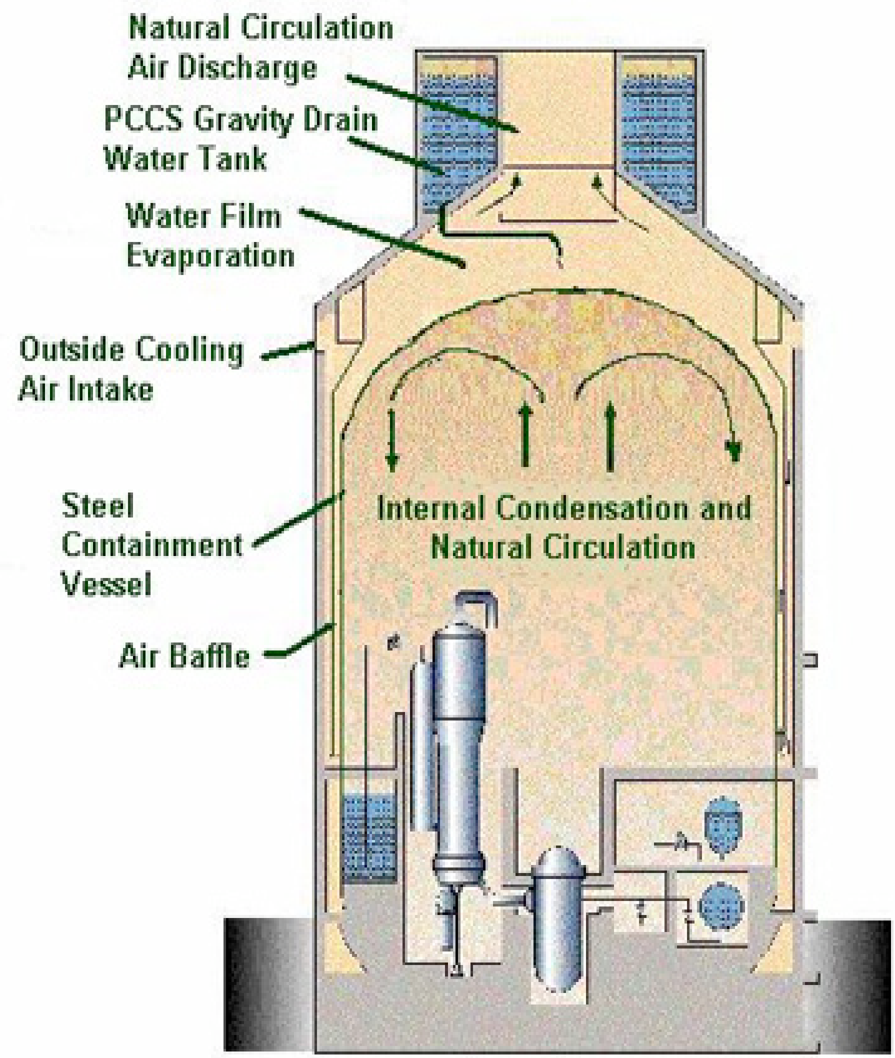
1950 1960 1970 1980 1990 2000 2010 2020 2030

Advanced Nuclear Reactor Systems

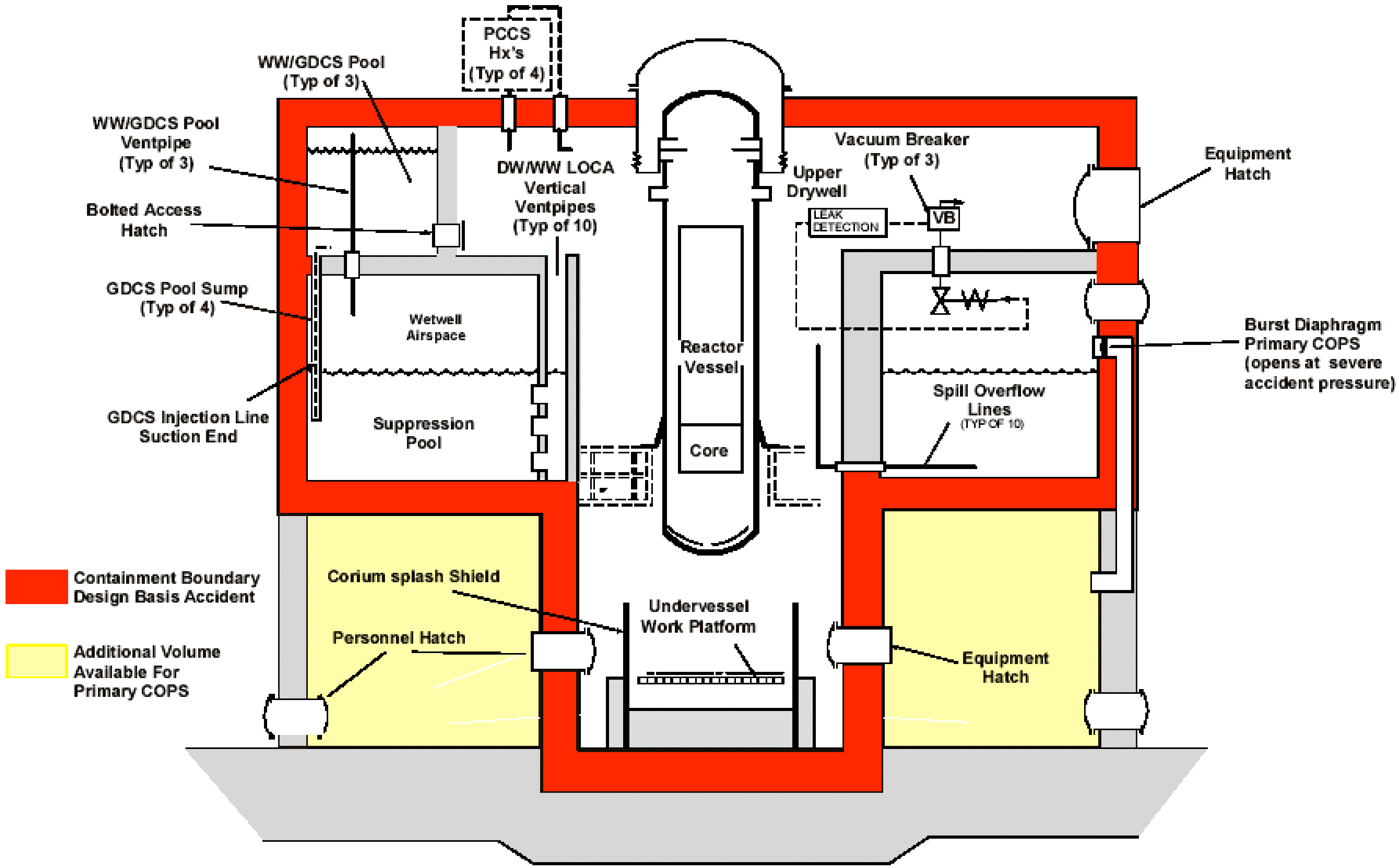
- Safety: meet and exceed current nuclear power plant reliability, occupational radiation exposure and risk of accident consequences
- Sustainability: minimize waste streams during fuel processing and spent fuel recycling and/or disposal
- Optimize physical protection of facility and non-proliferation risks
- Economics: reduce the total cost of electricity to remain competitive with other baseload power technologies (e.g., fossil fuels)

Advanced LWR: AP-1000

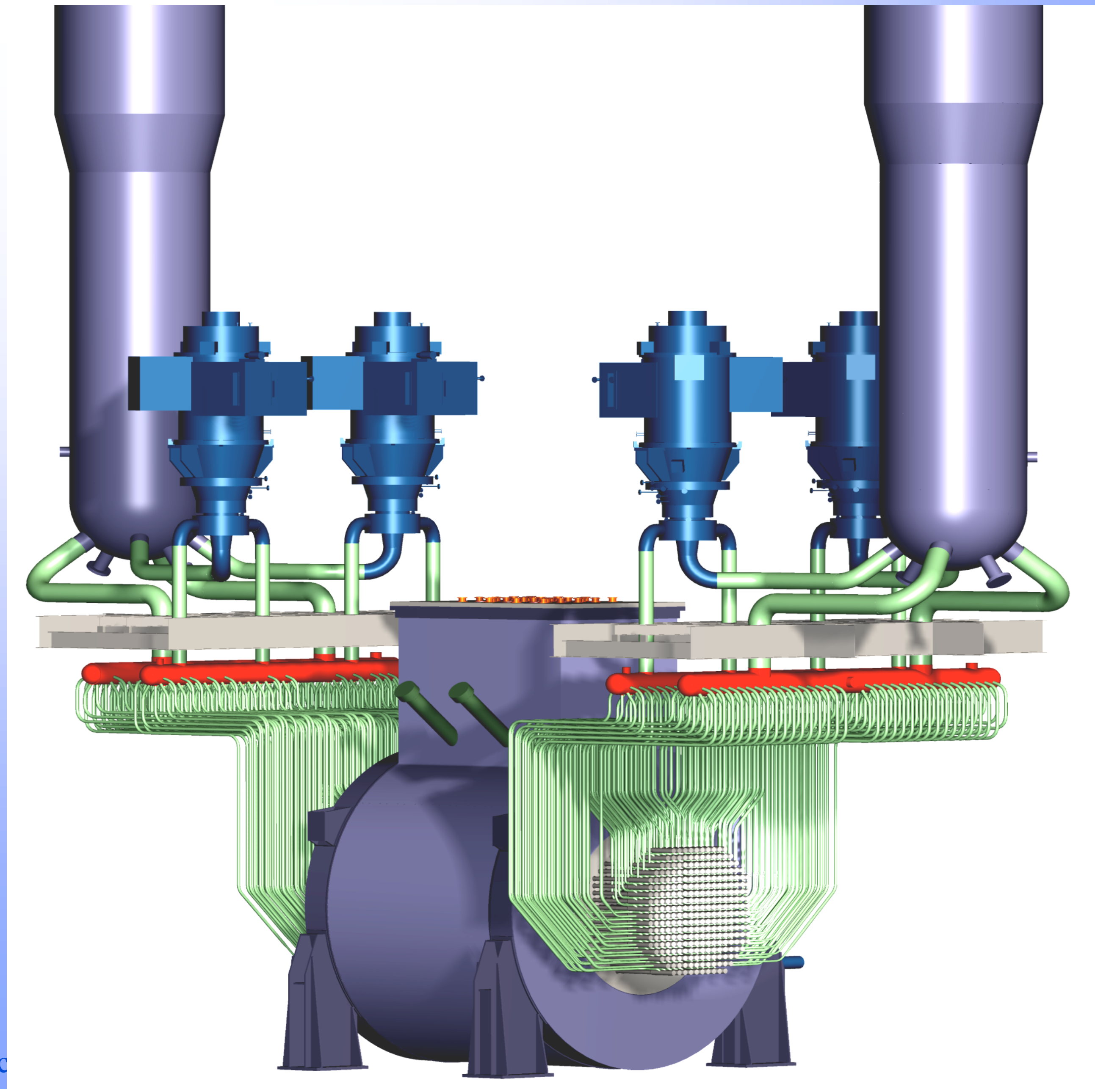


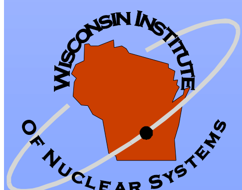


Advanced LWR: ESBWR

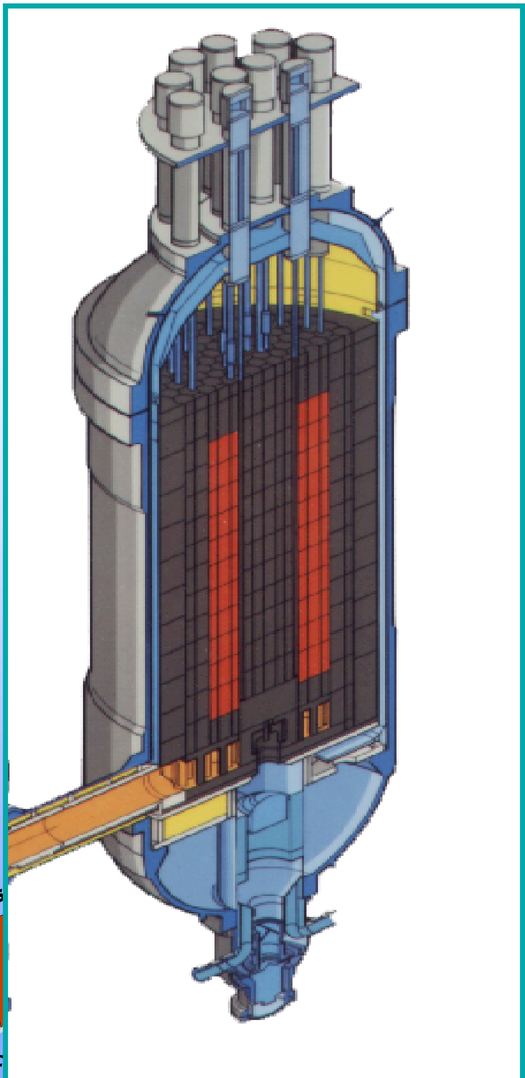


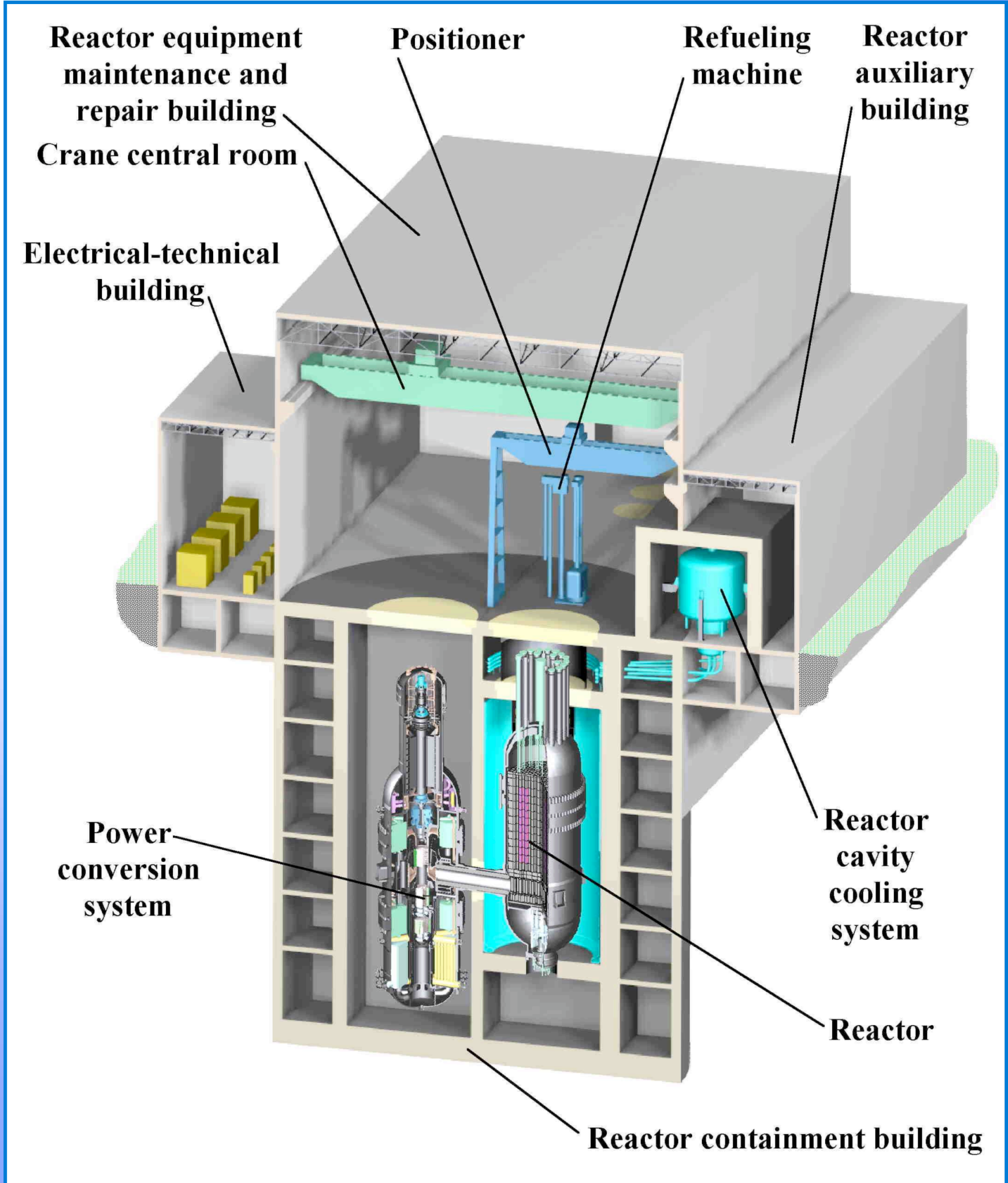
ACR-700





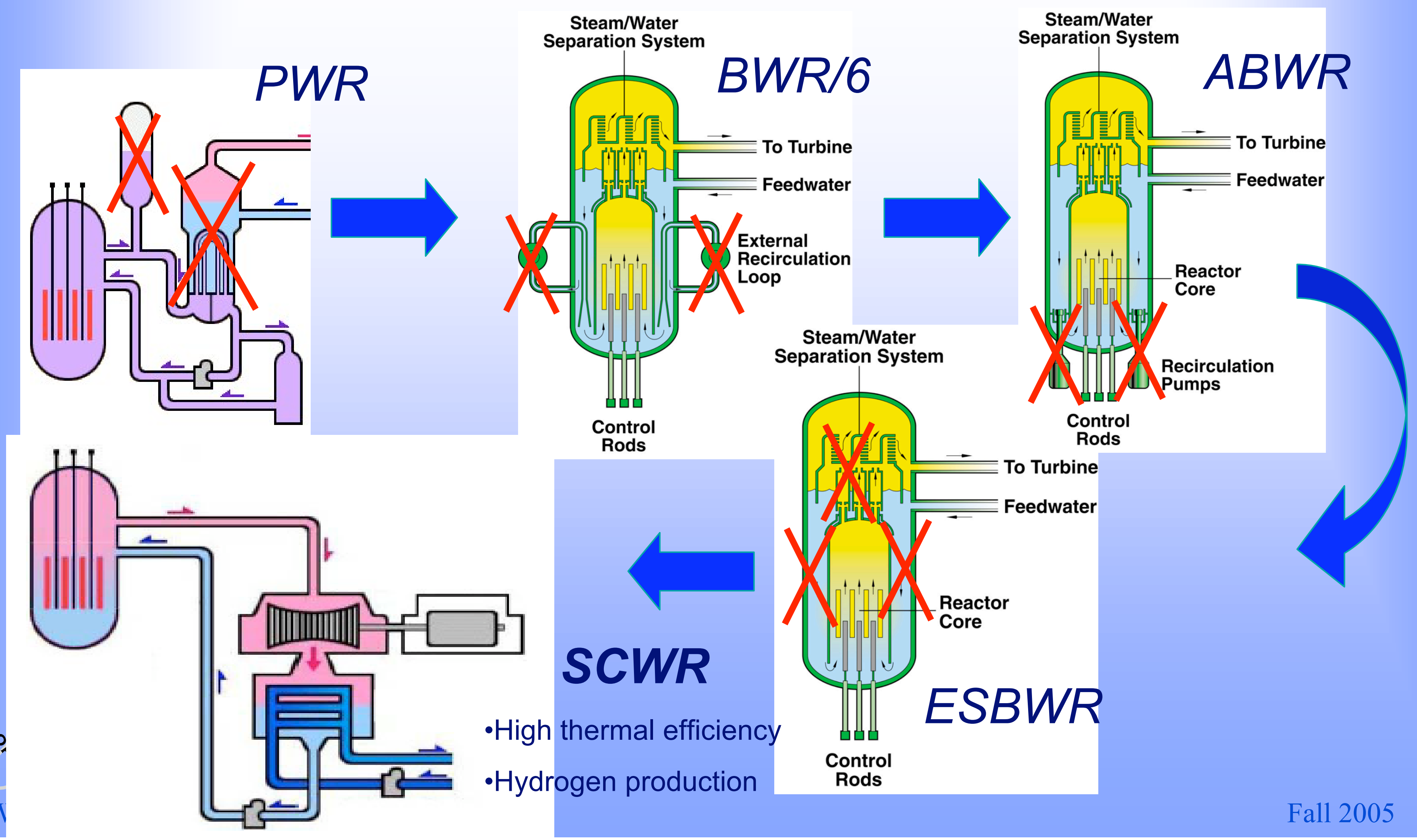
Advanced GCR
PBMR, MHTGR





SCWR: Gen-IV LWR

The next logical step in path toward simplification

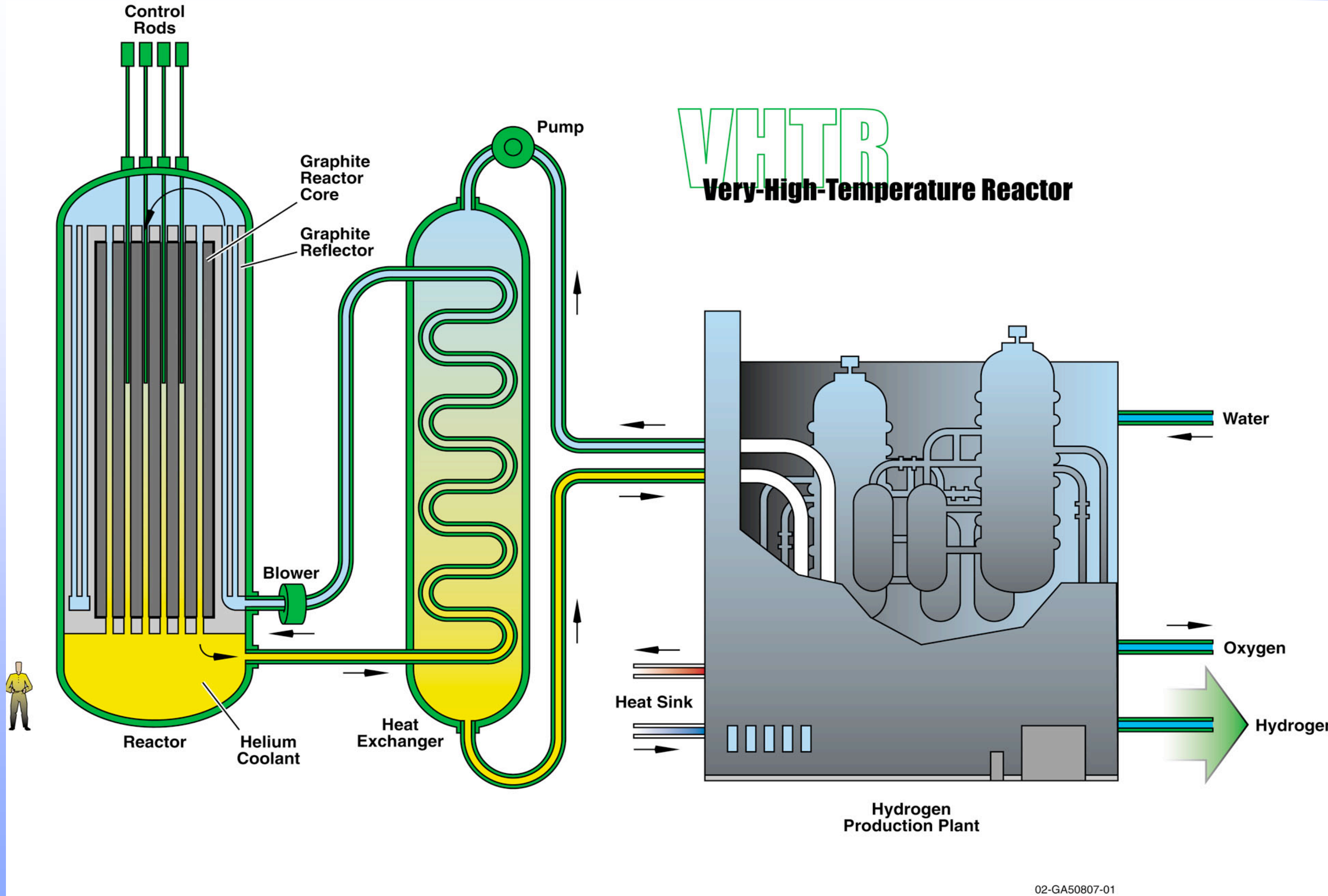


•High thermal efficiency

•Hydrogen production

Very-High-Temperature Reactor (VHTR)

- Characteristics
  - Helium coolant
  - 1000°C outlet temp.
  - 600 MWth
  - Water-cracking cycle
- Key Benefit
  - High thermal efficiency
  - Hydrogen production by water-cracking

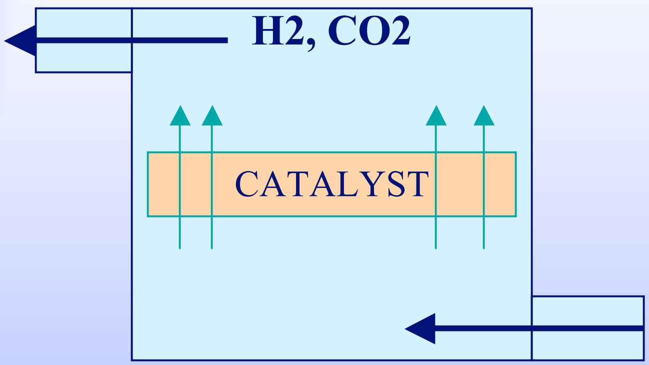


Process Heat for Hydrogen Production

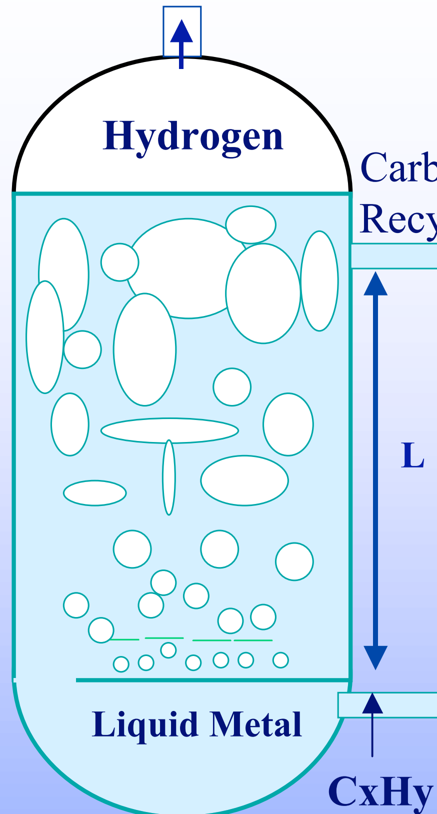
200 C

1000 C

Aqueous-phase Carbohydrate Reforming (ACR)

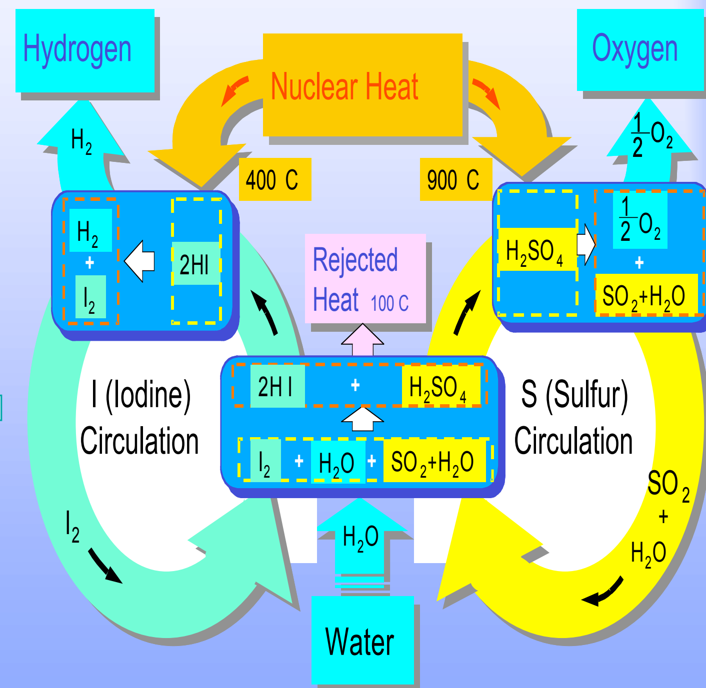


AQUEOUS CARBOHYDRATE



LM Condensed Phase Reforming (pyrolysis)

Thermochemical Processes



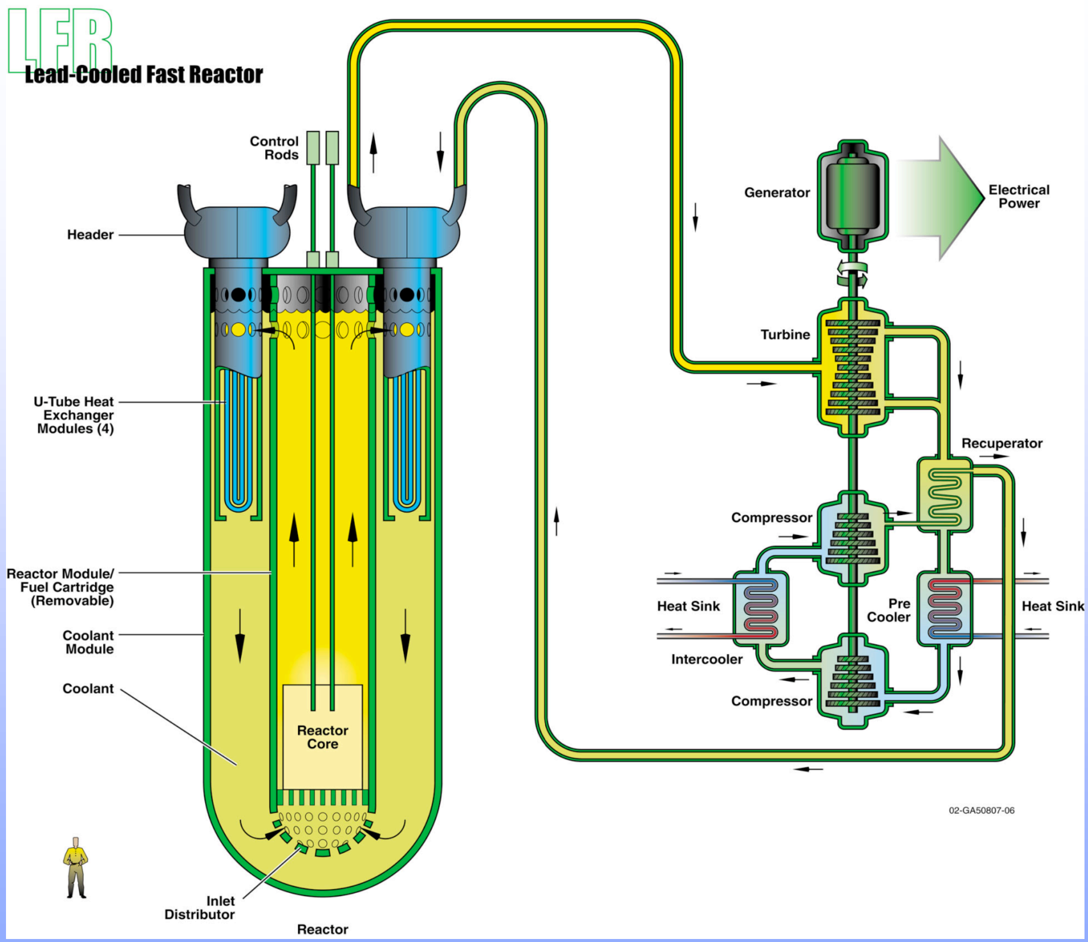
Lead-Cooled Fast Reactor (LFR)

Characteristics

- *Pb or Pb/Bi coolant*
- *550°C to 800°C outlet temperature*
- *120–400 MWe*

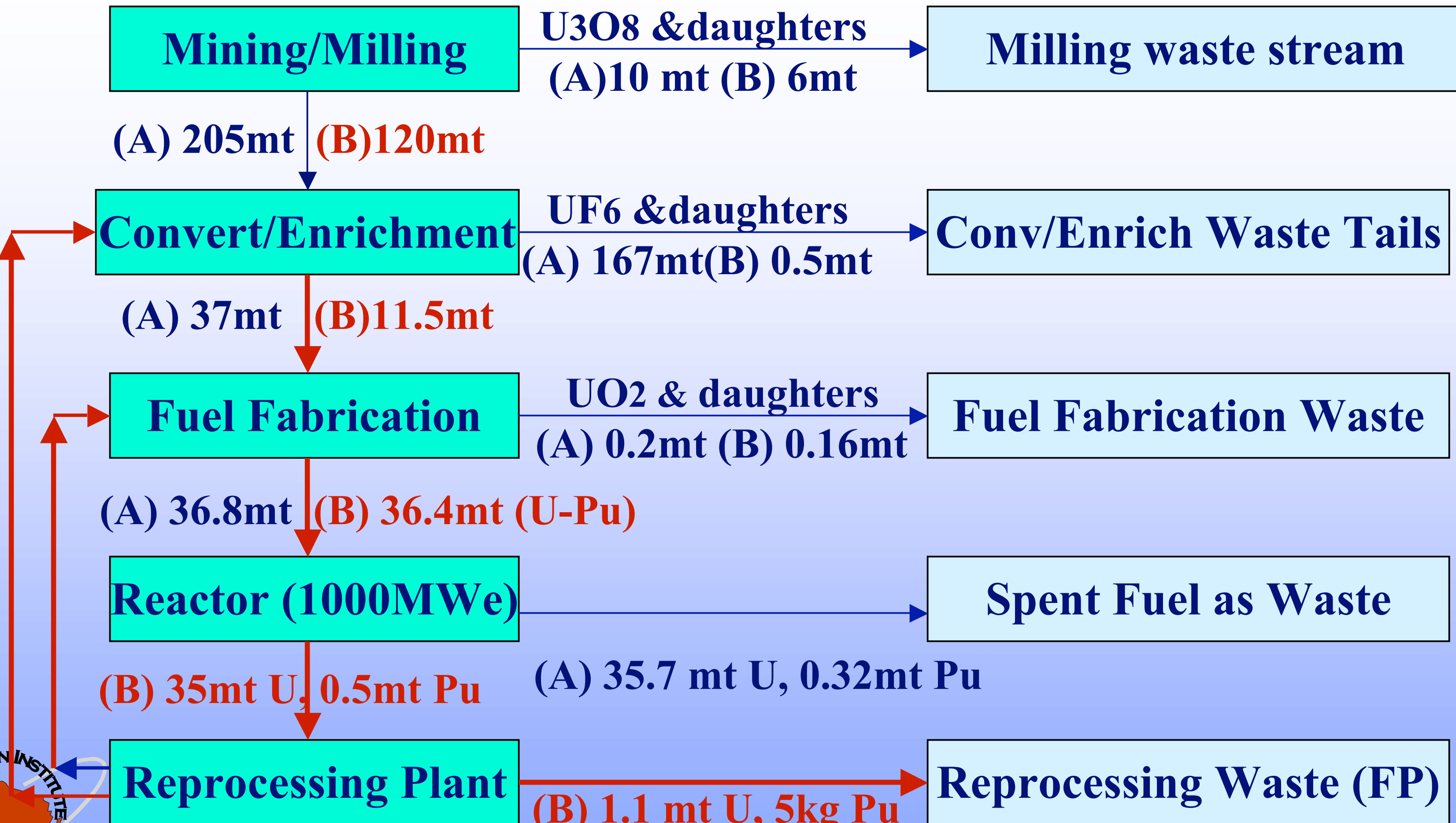
Key Benefit

- ***Waste minimization and efficient use of uranium resources***



Nuclear Power Fuel Cycle

[1GWe-yr – (A) Once Through (B) With Recycle; 3.3%U235, 30GWD/mt]



Advanced Reactors Regulatory Issues

Based on SECY-05-0130, NRC SRM 9-12-05, ACRS Ltr. 9-21-05

- ‘Technology Neutral Regulatory Framework’ is currently under development by the USNRC staff with ACRS input.
- NUREG-0880 Reactor Safety Goals are to be used as overall guidance (qualitative goals and quantitative health objectives).
- In the interim surrogate regulatory guidance follows approach for ALWR’s: i.e., DBA analyses and CDF & LER goals
  - DBA: Design Basis Accidents - Power-cooling mismatch events
  - CDF: Core Damage Frequency << 1/10,000 (PRA analyses)
  - LER: Large Early Radioactivity Release < 1/10 (w core damage)
- Usage of PIRT (Phenomena Ident. & Rank. Table) as a way to determine key issues needed for modeling and testing

Overview of PIRT Approach

1. Gather information and select Figures of Merit
2. Identify Scenario(s) to be Addressed for Review
3. Develop/Define Event Tree and the Phases for Scenarios
4. Identify Systems & Components Active During Scenario (by Phase)
5. Rank Systems & Components Active During Scenario (by Phase)
6. Identify Key Phenomena for Reactor System and Rank (by Phase & Component)
7. Identify the Key Issues for Phenomena
8. Compile Results and Document

PIRT Iterative Ranking Process

ACR-700 Example: Severe Accident Panel

SA Member	SA Scenario	SA Activity
M. Corradini	Single Channel	Evt.Tree, PIRT
S. Levy	Single Channel	Scenario, PIRT
R. Henry	Whole Core	Evt. Tree, PIRT
K. Vierow	Whole Core	Scenario, PIRT
D. Powers	Fission Prod.	Phen., PIRT

SEVERE ACCIDENT FIGURES of MERIT

- **Single channel events with limited core damage that do not propagate and degrade to a whole core accident**

- **Whole core accidents that achieve core debris coolability (in-vessel or ex-vessel)**

- **Prevent the release of radioactivity from containment from these (other) scenarios**

SA Event Scenario (example)

Table 2.1 Scenario and phase descriptions

(Single Channel Event Sequence: PT Strain Localization + Loss of Class III power)

Phase	Timing	General Phase Boundaries	Significant Events
I	0-30 sec.	**Fuel Channel Failure**	**Pressure Tube Failure** 1. Pressure Tube Failure (refer to event description). Non-uniform circumferential temperature distribution results in PT failure due to strain. 2. Pressurization of annulus between PT and CT up to the HTS pressure. 3. Water hammer pulse in annulus. 4. Subsequent bellows failure at both ends of the calandria tube 5. LOCA through both channel bellows **Plant Response Prior to CT Failure** 6. No reactor trip, assuming affected channel is not instrumented 7. Nominal conditions maintained by Pressure and Inventory Control System 8. Reactor Power maintained by Reactor Regulating System **Calandria Tube Failure** 9. Molten and solid fuel element material ejected to calandria tube 10. Transition to stratified flow pattern in calandria tube 11. Reduced cooling of top fuel elements 12. Melt relocation and contact with calandria channel 13. Calandria tube thinning at full pressure (Ref. 16, Figure 4-3) 14. Calandria tube failure For complete flow blockage PT/CT failure would happen in 10-12 seconds. For partial flow blockage it could take 40-60 seconds (ref. 5, Tables 7.1-5 and 7.1-6). **Plant Response after CT Failure** 15. HTS LOCA on the order of 100 kg/s 16. Reactor trip due to moderator high level, RCS (Reactor Cooling System) lower pressure, and pressurizer reduced level 17. Turbine trip (Timing per "LOCA due to 25% RIH (Reactor Header Inlet) Break with Subsequent Loss of Class IV Power"

Single Channel

ACR-700 Initial Conditions:
e.g., Peak Channel Power, Max Pressure Tube creep (4.5%); Channel not instrumented

LOCAL PWR/FLOW MISMATCH INITIATORS:
(i.e., Flow stagnation, Flow bypass, Flow blockage)

Thermal-Hydraulic transient detected

yes

Detect wet CO_2 or flow stagnation or void causing Rx scram: Accident termination

no

FUEL BUNDLE DEGRADATION
- Coolant voiding / liquid stratification
- Fuel/clad heatup and degradation
- Fuel/clad melting and mat'l motion

PT Failure: Local creep due to T>650C and/or melt contact

no

PT or CT do not fail and coolability is achieved

Yes AND

PT failure leads to PT Bellows failure & small break scenario; could lead to Calandria drain

Yes AND

CT Failure due to hot steam and melt contact
(note failure is assumed here)

MFMI FORCED INTERACTION: limited data or scaling available on propagation

C-Tube fails w Blowdown/Impact on CRDs/Tubes
- Pipe whip and collapse of other tubes
- Two-phase/fuel-clad mat'l dispersal
- Bubble expansion dyn. & level swell

Propagation to other channels and Calandria Failure

no

CALANDRIA SURVIVES
Limited damage
Rx scram
H2 Generation
Characterize debris

Accident Output Characteristics (FP release & transport, shutdown margin, H2 generation & transport)

yes

CALANDRIA FAILS
Degrades to whole core
Rx scram
FCI phen/H2 generation
Debris transport

Accident Output Characteristics (go to whole core severe accident sequence with Calandria failed)

Whole Core

ACR-700 Initial Conditions:
e.g., Peak Channel Power, Max Pressure Tube creep prior to accident

SEVERE ACCIDENT INITIATORS:
Required: Power/cooling mismatch with multiple heat sink failures
Initiators: LOCA (feeders, headers) and Station Blackout w Rx scram

Natural circulation & depressurize naturally at High-Pressure

no

Several PT fail w high-pressure melt ejection w several P-Tubes possible? (Probably near top of the Calandria)

This accident path similar to a MFMI phenomena

yes

Coolant stratification, voiding and instabilities with fuel heating

Fuel/clad melting in pressure tube

Pressure/Calandria tube failure if not failed at Hi -Pressure Cond.

Molten core debris relocation

FCI Fails Calandria

no

YES: NB 'Dynamic failure of Cal/ShTk/Cont sep. issue'

Debris coolablity on Calandria wall with a continuous water supply path

NO: NB 'Dynamic failure of Cal/ShTk/Cont sep. issue'

Debris coolability on Shield tank wall with a continuous water supply path

yes

Stable Coolable State achieved

no

Ex-vessel melt phenomena:
- ? Ex-vessel FCI
- ? Molten core-conc. Int.
- ? Hydrogen (rate/magn.)
- ? Fission prod. Release
- ? Ex-vessel coolability

yes

Stable State Final Condition

Containment leak, - Containment challenged – H2 Management

PIRT: Single Channel Accident Key Phenomena

Issue (Phenomena, process, geometry condition)	Importance for ACR-700	Rationale	Level of Knowledge	Rationale	Status of Severe Accident Modeling
Melt progression through pressure tube and calandria	High	Initial and long-term progression will fail pressure tube and calandria tube allowing fuel relocation downward amongst other tubes	Low	Extended melt progression information is probably not well-characterized in comparison with data base for melt progression in LWRs	Modification needed for SA codes to model this unique configuration
Pressurized expulsion of melt from the pressure tube into calandria	High	This is the key phenomena that may take a single channel event and propagate to whole core event	Low	This is an active area of experimental research by AECL to consider forced FCI interaction mode with chemical augmentation	AECL has stand-alone parametric unqualified model; may need a mechanistic model to provide scaling of loads and energetics.
Dry Core Melt Progression	High	High zirconium content in the molten material that is produced and moves due to slumping may directly cause Calandria and Shield tank failure	Low	LWR core melt contains a much lower amount of unoxidixed Zr compared to what may be in ACR-700	Needs discussion
Flow paths, flow splits and flow instabilities during severe accident progression	High	Flow paths dictate the ability to remove heat and to carry fission products through the reactor coolant system and into containment or, in the case of bypass accident sequences, to environment	Low	Complicated geometry of CANDU system leads to uncertain flow splits in parallel flow piping, with possible instabilities and additional PT failures and complex flow patterns to consider	Modifications to current severe accident computer models will be necessary to account for complex flow paths

PIRT: Whole Core Accident Key Phenomena

Issue (Phenomena, process, geometry condition)	Importance for ACR-700	Rationale	Level of Knowledge	Rationale	Status of Severe Accident Modeling
Melt progression through pressure tube and calandria	High	Initial and long-term progression will fail pressure tube and calandria tube allowing fuel relocation downward amongst other tubes	Low	Extended melt progression information is probably not well-characterized in comparison with data base for melt progression in LWRs	Modification needed for SA codes to model this unique configuration
Creep of pressure tubes during whole core degradation	High	Pressure tube creep affects cooling and can bring Zr tubes into contact with calandria tube	Low	Limited data base on heat transfer from creeping tubes during whole core degradation	Major modifications

ACR 700 Key Issues and Approach

- Severe Accident PIRT process concluded with identification of key phenomena of high priority
  - Core melt progression with neutronic feedbacks
  - Pressurized expulsion of melt w PT/CT failure
  - Pressure tube creep rupture during whole core event
  - Flow paths, flow splits, flow instabilities in accident
  - Dry-core melt progression and debris coolability
- Future safety research needs to address modeling and experimental knowledge base needed to meet goal
- ***Focus on passive safety and longer time for response***

Advanced Reactor Safety Research

- Current NRC's advanced reactor research applies principally to certain reactors: AP1000, ACR-700, ESBWR, PBMR, GT-MHR and IRIS. There are several key research areas:
  - Neutral regulatory framework (regulatory decision-making based on the risk-informed, performance-based principles)
  - Improved techniques for accident analysis (e.g., PRA methods and assessments, human factors, and instrumentation and control)
  - System models (e.g., TH analysis, nuclear, severe accident and source term analysis)
  - Advanced fuels analysis and associated testing
  - Materials analysis (e.g., graphite behavior and high-temperature metal performance)
  - Structural analysis (e.g., containment/confinement performance and external challenges)
  - Consequence analysis (e.g., dose calculations, and environmental impact studies)
  - Nuclear materials safety (e.g., enrichment, fabrication, and transport) and waste safety (including storage, transport, and disposal), and nuclear safeguards

Reactor Safety Research Issue Matrix

Research Area	Advanced Water Reac.	Gas Cooled Reactors	Liquid Metal Reactors	Hi-Perform. Computing
PRA analysis - assessment	Improve techniques to allow for technology neutral assessments, analysis & consequences			PRA techniques e.g., ROAAM, MELCOR
Reac. system analyses	P-TH transients Core coolability	Mod. response temp & radiation	Failure P-P prop Trans. O-P anal.	Neutronics-TH coupled anal.
Materials analysis	Hi-Temp Corros. & Mat'l Damage	Graphite prop. Surf. Emissivity	Fatigue Failure Fuel Parameters	Computational Mat'ls & Props
Structural analysis	High-temp. creep behavior	Heat exchanger struct'l. integrity	Fuel and core support analysis	Fluid-Structure coupled analy.
Consequence analysis	Fission product release and transport is dependent upon failure mechanisms and local chemistry.			Fission product transport

Reactor Safety Research: ALWR's

Current NRC's advanced reactor research applies to certain water reactors: AP1000, ACR700, ESBWR and IRIS. Examples include:

- System power/temperature response to modifications in LWR operating conditions and geometry:
  - ESBWR: Condensation heat transfer and mixing PCCS
  - ACR700: Void and temperature coefficients in ACR geometry
  - IRIS: System TH analysis given design-basis accident initiators
  - SCWR: Heat transfer deterioration near pseudo-critical point

⇒ HPC initiative in neutronics/thermal-hydraulics coupled models

- Debris coolability in-vessel (or ex-vessel) for specific designs
- Creep and creep-fatigue in design and safety computer models

Reactor Safety Research: GCR's

Current NRC's advanced reactor research applies to certain water reactors: PBMR and MGTHR. Examples include:

- T-H system analyses for LOF & LOP accidents with air ingress (this is the analogue to water reactor design basis and beyond)
- Graphite swelling from fluence & temperature variations in core:

⇒ HPC initiative in coupled neutronics/heat-transfer effects

⇒ HPC initiative in first-principles materials properties

- Emissivity-by-design: passive surface cooling of RPV in accident

=> HPC initiative with testing in stable surface props (temp. & rad.)

- Effect of mixed-oxides and actinides on neutronics safety parameters: delayed neutron fraction, Doppler feedback, thermal conductivity, etc. => HPC initiative on fuel properties

Reactor Safety Research: LMR’s

Current NRC's advanced reactor research applies to certain water reactors: SFR’s and LFR’s. Examples include:

- ◆ T-H system analyses for transient overpower and LOF/LOHS accidents as well as pin-to-pin propagation failures

⇒ HPC initiative in first-principles multi-dimensional fluid dynamics

⇒ HPC initiative in coupled neutronics/heat-transfer effects

- ◆ Effect of mixed-oxides and actinides on neutronics safety parameters: delayed neutron fraction, doppler feedback, thermal conductivity etc.

=> HPC initiative on fuel properties as a function of fissile composition as well as fission product and minor actinide content

Hi-Performance Computing Focus

Consider now the common attributes from all of these examples for various advanced reactor designs and associated accident scenarios:

- As computer modeling capabilities become more sophisticated the tools used for design and safety will become "one and the same".
- As these fields continue to merge => design-to-analysis capability will also lead to direct interface between CAD and high-fidelity coupled multi-physics capabilities (neutronics+TH+fuel performance+structural analysis+..)
- Imagine reactor system analysis with Monte Carlo: simplified temperature-dependent analysis with coupling to other physics (TH + Fuel + Structures)